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Effects of Liner Degradation on the Severe Accident Consequences at a PWR Plant with a Reinforced Concrete Containment

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1 ABSTRACT

Since degradation has been observed in the containment vessels of a number of operating nuclear power plants in the United States, a number of studies have examined the increase in the risk of a fission product release in the unlikely event of a severe accident. Here, deterministic analyses were performed in the current study to assess the effects of degradation on the consequences. These analyses were performed using the Sandia codes MELCOR and MACCS. These codes were used to simulate two different accident scenarios (long- and short-term station black-outs) and compute the resulting consequences for a pressurized water reactor plant with a reinforced concrete containment. The structural analyses were used to develop the containment behavior models for the accident simulations. Several different postulated cases of liner corrosion were considered to enable a comparison of the consequence.

2 INTRODUCTION

Various forms of degradation have been observed in the containment vessels of a number of operating nuclear power plants in the United States. Examples of degradation include corrosion of the steel shell or liner, corrosion of reinforcing bars and prestressing tendons, loss of prestressing, and corrosion of bellows. The containment serves as the ultimate barrier against the release of radioactive material into the environment. Because of this role, compromising the containment could increase the risk of a release in the unlikely event of a severe accident.

Previous work (Cherry et al., 2001) in this area has assessed the effects that degradation has on the pressure retaining capacity of the containment vessel. While such analyses have provided useful information about the effects of the degradation on the structural capacity of the containment, they did not necessarily provide a perspective on the effect that the degradation could have on the risk associated during a severe accident. Therefore, additional work conducted by Spencer et al. (2006) integrated structural analysis results with pre-existing probabilistic risk assessment (PRA) models in this study to gain a risk-informed perspective on the issue of containment degradation. These PRA models used were originally developed for NUREG-1150 (USNRC, 1990) and NUREG/CR-4551 (Breeding et al., 1992). The coupled structural-risk approach was applied to case studies of containment degradation at four “typical” U.S. nuclear power plants used in NUREG-1150. This approach enabled the determination of the risk, in terms of the large early release frequency (LERF), as well as the changes in LERF due to postulated cases of degradation. An instance of degradation was treated as a change in the plants license basis and assessed with U.S. NRC Regulatory Guide 1.174 (USNRC, 2001). The limits of the acceptable increases in LERF are provided in Reg. Guide 1.174.

The majority of the postulated cases of degradation considered by Spencer et al. (2006) consisted of local corrosion in the liner or shell which are typical in many existing nuclear containments. However, local corrosion in the liner of a typical reinforced or prestressed concrete containment was argued to only produce a noticeable effect on the pressure at which a leak would initiate, and not the pressure at which larger rupture or catastrophic rupture failures would occur. Degradation that only affects the leak pressure does not contribute to LERF due to the binning process used in the NUREG-1150 PRA models. In some cases, LERF was shown to decrease due to corrosion. These decreases in LERF result from the earlier small leaks due to local corrosion instead contributing to the small early release frequency (SERF). Regulatory Guide 1.174 does not provide guidance on the SERF limits, and therefore, may not be appropriate when assessing the risk significance of degradation in containment structures. Since LERF and SERF are only surrogates for the release consequences, additional deterministic source term and consequence analyses were performed in this study to assess the effects of degradation on the consequences beyond LERF.

The consequence analyses performed in this study employ two codes developed at Sandia National Laboratories, MELCOR and MACCS. MELCOR 1.8.6 (Gauntt et al., 2005) is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. MELCOR is under ongoing developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code Package. A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework. These include thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heat-up, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior. Current uses of MELCOR include estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications. The MACCS code (Chanin et al., 1990a, 1990b, and 1990c) was developed for the U.S. Nuclear Regulatory Commission (NRC) to evaluate offsite consequences of hypothetical severe accidents at nuclear power plants (NPPs) based on calculated or assumed fission product releases. The code is designed to take input directly from the fission product releases calculated by MELCOR. MACCS also incorporates geographic, demographic, and meteorological data for a given plant, as well as assumptions concerning biological uptake. The output results can be expressed in terms of health consequences. However, the present study only considers the health consequences in terms latent cancer fatality risk.

A PWR plant with a subatmospheric reinforced concrete containment was examined for both a long and short term station blackout accident. Both of these scenarios produce the high internal pressures necessary to initiate cracks in the steel liner and outer concrete containment wall. The degradation postulated for this containment was restricted to local corrosion in the steel liner near the wall-basemat junction and at the mid-height of the containment. The structural analyses that incorporate the effects of this degradation were completed during the previous study (Spencer et al., 2006) that employed the NUREG-1150 PRA models. Those results were used in this work to develop crack open area vs. internal pressure distributions for each of the postulated corrosion cases. These area vs. pressure curves were introduced into the containment representation within the MELCOR analyses. The resulting source term output for each case of degradation was then input into the MACCS code to determine the consequences. The changes in consequences due to differences in corrosion location and extent are compared to the analyses of the containment without degradation. Even though the structural models, MELCOR, and MACCS models were used for a specific plant, the cases of degradation assumed in this study are postulated in nature and do not reflect the actual conditions at any existing nuclear power plant. This study was conducted solely to demonstrate an analysis methodology and to examine the sensitivity of the consequences to varying cases of degradation.

3 REINFORCED CONCRETE CONTAINMENT STRUCTURAL ANALYSIS

A schematic diagram of a PWR with a reinforced concrete containment building is shown in Fig. 1 with the location of two postulated regions of liner corrosion. The reinforced concrete containment is approximately 1.37 m (4.5 feet) thick with a 9.53 mm (0.375 inch) thick steel liner. The axisymmetric finite element mesh of the containment used in the previous study by Spencer et al. (2006) is illustrated in Fig. 2. The concrete is modeled using the ANACAP-U (ANATECH, 1997) constitutive model, which is linked with the ABAQUS (Hibbet et al, 2005) finite element program. Reinforcing bars are modeled by embedding them within the concrete elements. The liner is modeled with axisymmetric shell elements directly attached to the concrete elements on the inside face of the containment. This example analysis adopts the baseline material properties used in the previous work by Spencer et al. (2006). These are the median (50%) values employed when introducing the structural analysis results into the PRA models described previously. The property parameters include the compressive strength of the concrete (27.3 MPa), the ultimate strength of the rebar (508 MPa), the ultimate strength of the steel liner (577 MPa) and the uniaxial failure strain (25%), among others.

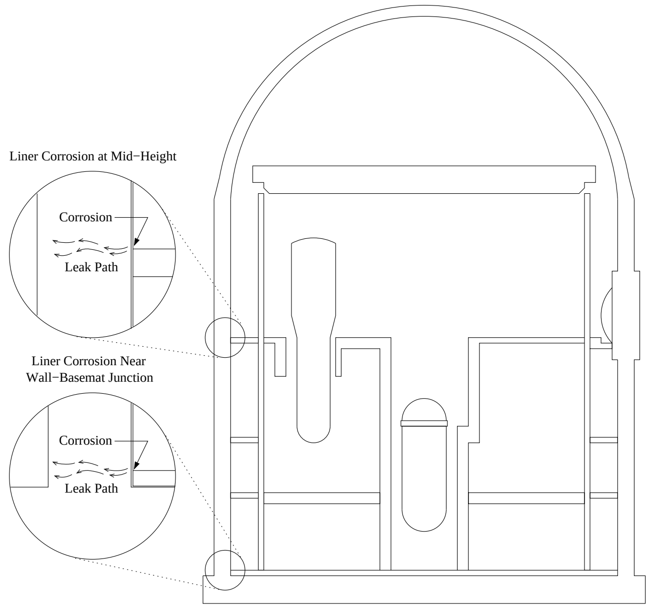

Figure 1. Reinforce Concrete Containment Leak Paths

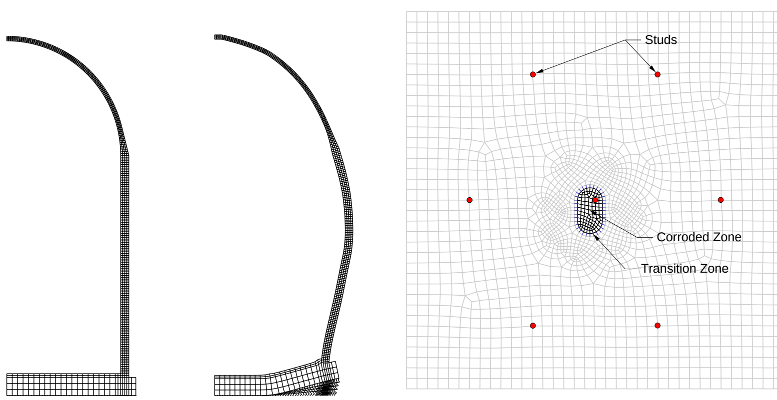

Figure 2. Reinforced Concrete Containment Finite Element Model

Nonlinear spring elements are used to approximate the contact conditions between the containment and the soil below. The spring elements have a high elastic stiffness in compression, and have no strength in tension. The containment model is subjected to a monotonically increasing internal pressure loading. Along with the undeformed model, Figure 2 shows the deformed shape of the axisymmetric model for a typical case near the end of the analysis. In addition, Figure 2 illustrates the detailed 2-D finite element models of the steel liner used to model regions of hypothetical corrosion at the mid-height and wall-basemat junction of the containment. Appropriate boundary conditions are extracted from the global axisymmetric analysis and are applied to the boundaries of the submodel. The mesh covers a 91 cm (36 in) by 91 cm (36 in) square region. The average length of an element edge is 25 mm (1 in), although, the mesh is refined considerably in the corroded area with an average element length of approximately 8 mm (0.3 in). In this corroded region, a single layer of elements surrounding the fully corroded region is used to make a more gradual transition from the un-damaged area to the corrosion zone. The effects of corrosion are introduced by reducing the thickness in that region of the submodel.

As the pressure increases, the containment will eventually begin to leak, and the leakage rate will increase with increasing pressure. It often proves challenging to translate structural response metrics such as strain and displacement into leak rates. Tears (leaks) in containment liners will first initiate in regions of stress concentration, which can be caused by welds, local geometric details, or corrosion. For the steel liners in concrete containments, it is assumed that the "leak" initiates when a tear occurs, since experience in testing large scale models of concrete containments has shown that the concrete cracks prior to tearing in the liner. Therefore, a pathway for material to exit the containment is assumed to exist at the onset of tearing in the steel liner. Spencer et al. (2006) employed a model where a tear is assumed to initiate when the a modified effective plastic strain, ε_p, exceeds the plastic strain found from a uniaxial tension test, denoted as $\varepsilon_{f\text{-}uniaxial}$. Since conditions within a real 3-D structure exhibit much more complex stress-strain states than uniaxial tension tests, the effective plastic strain, $\varepsilon_{p,eff}$, or global strain, ε_g, from the liner in a finite element analysis, is modified to adjust for these effects.

$$\varepsilon_p = f_m \, f_c \, f_g \, \varepsilon_{p,eff} \quad \text{or} \quad \varepsilon_p = M \;\; \varepsilon_g \tag{1}$$

The first part of eqn (1) shows the factors that are multiplied by the effective plastic strain, $\varepsilon_{p,eff}$, from the analysis, in the adjustment procedure to compute ε_p. The factors modify the effective plastic strain to include the effects of the multiaxial stress state (f_m), corrosion (f_c) if present, and for the gauge length (f_g). The first part of eqn (1) is used to predict tearing in the liner of concrete containments where corrosion is located in the detailed liner models. In regions of typical discontinuities and penetrations, we employed the second part of eqn (1) and the results of detailed analyses that were performed by Tang et al. (1995). In this study, strain magnification factors were developed to approximate the effects of discontinuities typical of those found in the liners of reinforced and prestressed concrete containments. Four locations have been identified as being critical for tearing: at large steam penetrations, at the junction of the wall and the basemat, at personnel or equipment hatches, and at the springline. For the reinforced concrete containment, Fig. 3 shows plots of these four magnification factors as functions of the normalized global strain. The normalized global strain, ε_g, is simply the appropriate global strain quantity divided by the yield strain of the liner material. The use of the second part of eqn (1) and Fig. 3 allows for performing axisymmetric analyses that do not include detailed models of the discontinuities. The global strain, ε_g, at the location in the finite element model where the specific discontinuity is located within the actual structure is multiplied by the corresponding magnification factor, *M*, in Fig. 3.

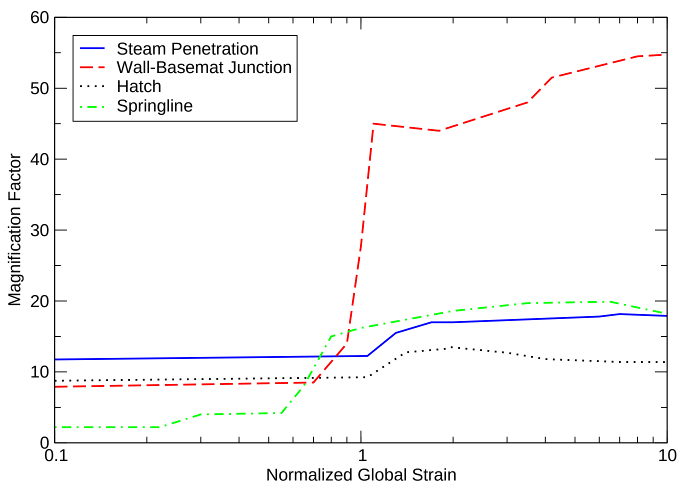

Figure 3. Strain Magnification Factors

For the four discontinuities in Fig. 3, and a case of postulated corrosion, the value of ε_p is computed as a function of the internal pressure. Since ε_p is an equivalent uniaxial strain, tearing occurs when it exceeds the plastic strain found from a uniaxial tension test, denoted as $\varepsilon_{f\text{-}uniaxial}$. At this point, a leak has formed at that location and at that level of loading (internal pressure). Since the analyses performed here do not explicitly model the actual crack (leak) initiation and growth, an approximate method must be employed to estimate the hole size. The method developed by Dameron et al. (1995) is used here for that purpose. Given a location of an initiated crack using the above criteria in eqn (1), the global strain, ε_g, at that location is used with the spacing between studs or anchorages to estimate the crack width and length in that region. These estimates are then used to compute an estimated crack opening area. Since cracks can exist simultaneously at multiple locations, the total crack opening area is computed by summing the contribution of all cracks throughout the containment at a given pressure.

In this example, 5 cases or scenarios are investigated: 1) the undegraded containment which includes the four discontinuities in Fig. 3, 2) 50% corrosion at the mid-height, 3) 65% corrosion at the mid-height, 4) 10 regions of 50% corrosion at the mid-height, and 5) 10 regions of 65% corrosion at the mid-height. The cases of near-basemat corrosion examined for the PRA analyses performed for the Spencer study were determined to not vary from the no corrosion case due to other failures dominating, and therefore, were not examined. The percentage of corrosion indicates the amount of the original steel liner that is uniformly removed from the original 9.53 mm (0.375 inches) thickness (t) (50% corrosion: t = 4.76 mm, 65% corrosion: t = 3.33 mm). For cases 4 and 5, 10 regions of corrosion are assumed to exist within the steel liner. Each of these 10 cases are assumed to be identical to each other but are assumed to be structural independent. Therefore, the crack open area computed from the analysis of one region (shown in Fig. 2), was simply multiplies by 10 and added to the crack open area for the four discontinuities in Fig. 3. No cases were examined that include corrosion at both the wall-basemat and mid-height locations simultaneously. The crack opening area associated with corrosion for each of the cases 2 through 5 are essentially added to the base undegraded case 1. It is assumed that these are the points of the earliest (lowest pressure) possible leaks in the containment, and that other potential leaks would occur only at higher pressures.

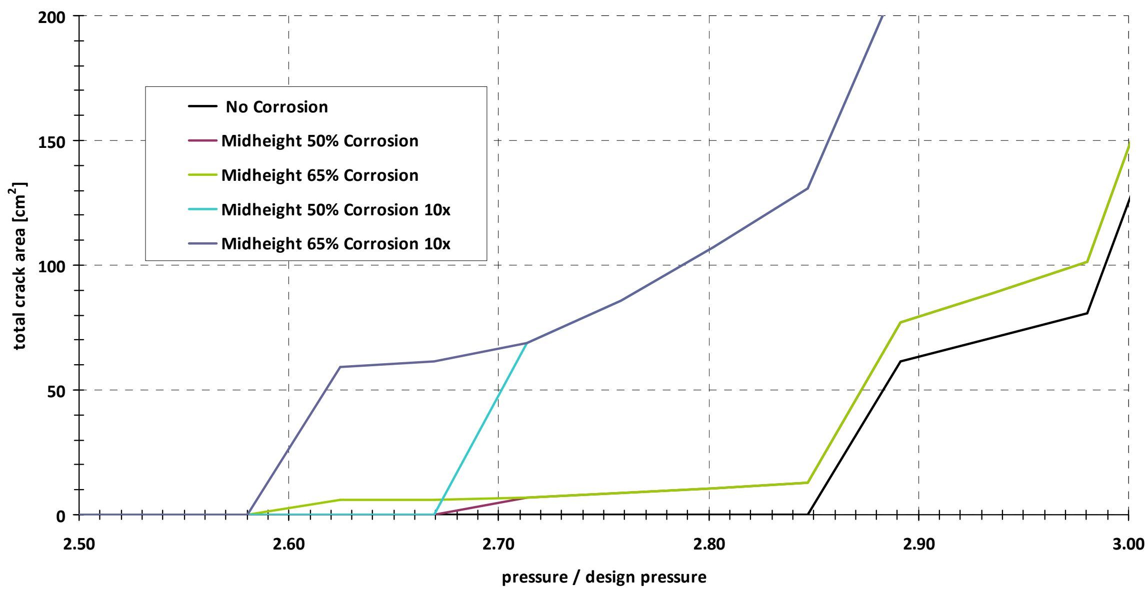

Figure 4. Crack Area vs Pressure

Figure 4 illustrates the area vs. normalized pressure curves for the 5 cases described above. The normalized pressure is the pressure divided by the design pressure for the containment. The cases with corrosion at the mid-height produce a significant effect on the total crack area. This is due to the higher strains at the mid-height of the containment leading to the computation of large cracks. The curves for the cases with 50% and 65% corrosion at the mid-height lie slightly higher than the case with no degradation, with the 65% corrosion case initiating at a slightly lower pressure the for 50% corrosion. In order to explore cases of additional corrosion, the introduction of 10 regions of 50% or 65% corrosion produce much larger total crack area curves than the case with no degradation. The modelling here shows that the initiation of through-thickness cracks do not occur until the internal pressure exceeds 2.5 times the design pressure. The jumps in the crack area curves in Fig. 4 are the result of the initiation of a crack at a new location (e.g. corrosion location or one of the four discontinuities in Fig. 3).

4 MELCOR SEVERE ACCIDENT ANALYSES

MELCOR 1.8.6 (Gauntt, et al., 2005) is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code Package. A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework. These include:

- thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings;
- core heatup, degradation, and relocation;
- core-concrete attack;
- hydrogen production, transport, and combustion;
- fission product release and transport behavior.

Current uses of MELCOR include estimation of severe accident source terms and their sensitivities and uncertainties in a variety of applications. The MELCOR Users Guide (Gauntt et al., 2005) provides extensive descriptions of the codes capabilities.

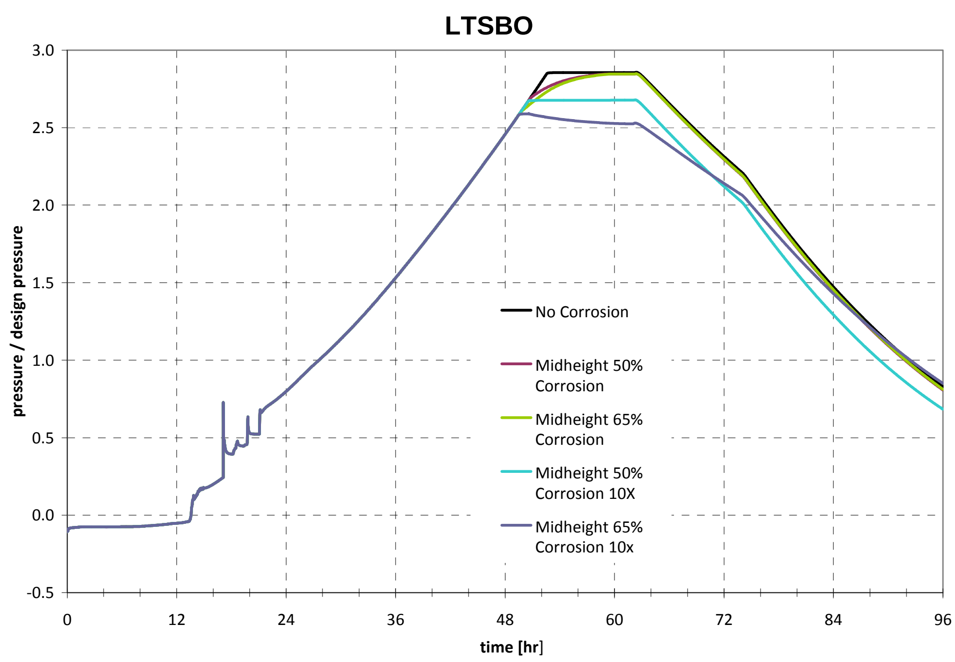

Figure 5. LTSBO Containment Pressure vs Time

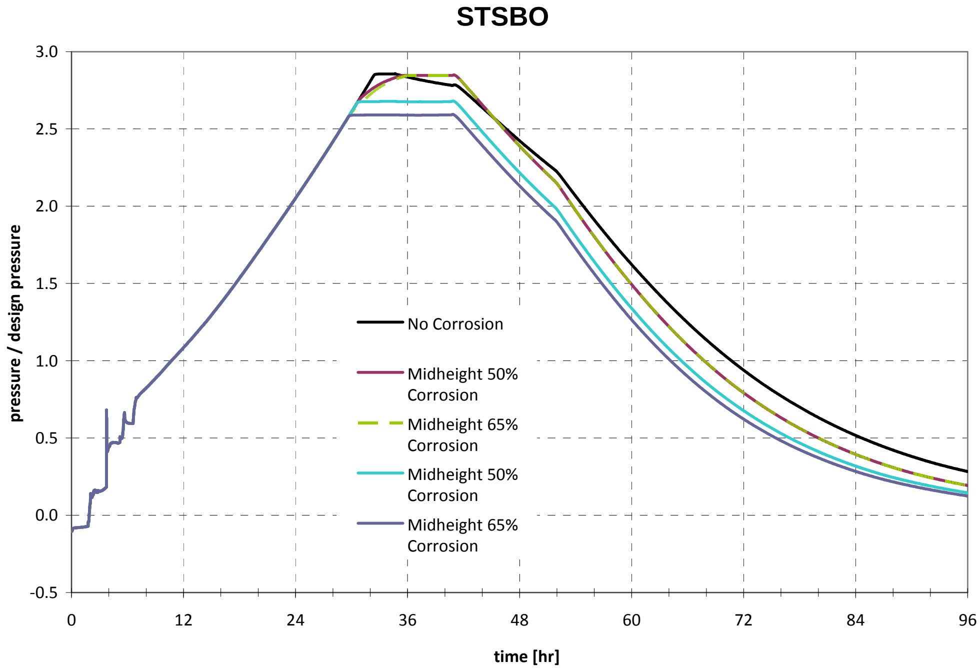

Figure 6. STSBO Containment Pressure vs Time

A MELCOR model for this type of plant has been developed by Sandia for the purpose of testing new models, advancing the state of the art in modelling of PWR accident progression, and providing support to decision makers at the U.S. Nuclear Regulatory Commission (NRC) for analyses of various issues that may affect operational safety. Calculations were performed for both the short-term station black out (STSBO) and long-term station black out (LTSBO) scenarios. Detailed accident progression sequences of events for both scenarios using the containment failure models were employed. The STSBO accident begins with a loss of all onsite and offsite power, however, the LTSBO includes battery a backup for a specified interval before exhausting power. Both accident scenarios progress through pressurization of the containment vessel and release of fission products through the resulting cracks in the containment vessel.

The 5 containment scenarios were evaluated for both the STSBO and LTSBO accidents. Figures 5 and 6 show the normalized containment pressure (normalized by design pressure) as a function of time for each corrosion case. The release constituents are also computed in MELCOR for each of these cases to be used as input in the MACCS analyses in the next section.

5 CONSEQUENCE ANALYSES

The MACCS2 code (Chanin et al., 1990a, 1990b, and 1990c) was developed for the U.S. Nuclear Regulatory Commission (NRC) to evaluate offsite consequences of hypothetical severe accidents at nuclear power plants (NPPs) based on calculated or assumed fission product releases. The code is designed to take input directly from the fission product releases calculated by MELCOR. MACCS2 models the radioactive materials being released as being dispersed in the atmosphere while being transported by the prevailing wind. Downwind population would be exposed to radiation, and land, buildings, and crops would be exposed to radioactive materials deposited from the plume. MACCS2 estimates the range and probability of the health effects induced by the radiation exposures. More detailed information on MACCS2 can also be found in the code manual for MACCS2 (Chanin et al., 1990a, 1990b, 1990c).

The offsite consequence analysis was performed using MACCS2 for each of the source terms generated by cases examined in MELCOR. The MACCS2 input model employs site specific emergency response scenarios for severe accidents that are characterized with the timing of actions taken by onsite and offsite response organizations to protect the general public by evacuation and sheltering. These emergency response scenarios were developed in order to achieve a greater degree of realism than previous consequence analyses performed by MACCS2. However, this study is being used to demonstrate a methodology and should not be considered site specific.

The following section presents MACCS2 consequence results for LTSBO and STSBO scenarios. While MACCS2 consequence model calculates a large number of different consequence measures including total latent cancer fatalities, population dose, and latent cancer fatality risk, only the normalized latent cancer fatality risk values are reported here. It should be noted that the results presented here are based on linear-no-threshold hypothesis (LNTH), where an assumption is made that the relationship between exposure to ionizing radiation and human cancer risk is linear. The LNTH model is simply chosen for simplicity in setting up the problem and easier comparison of the results between different runs.

A total of 5 different corrosion conditions were evaluated for both STSBO and LTSBO scenarios at 96 hours after the start of the accident. MACCS2 generated output in terms of probability distributions of the consequence estimates resulted from the statistical variability of seasonal and meteorological conditions during the accident. While the results produced with the site specific meteorological data addressed uncertainty in the weather, it was difficult to draw conclusions from this set of results due to abnormalities observed in some cases. Because of this, it was decided that the cases be re-analyzed using only a single weather trial so that the results produced are more deterministic in nature. Results obtained using single weather trial are those presented in Tables 1 and 2.

Table 1. STSBO Risk.

Corrosion Case	**Individual Latent Cancer Fatality Risk, 16 km (Percent of No Corrosion Case)**
Midheight 50% Corrosion	123.8%
Midheight 65% Corrosion	123.8%
Midheight 50% Corrosion 10x	170.3%
Midheight 65% Corrosion 10x	161.3%

Table 2. LTSBO Risk.

Corrosion Case	**Individual Latent Cancer Fatality Risk, 16 km (Percent of No Corrosion Case)**
Midheight 50% Corrosion	103.3%
Midheight 65% Corrosion	103.3%
Midheight 50% Corrosion 10x	122.1%
Midheight 65% Corrosion 10x	115.3%

It was anticipated that consequences (e.g., risk) would be proportional to the extent of the corrosion. The logic behind this being that more corrosion would yield larger containment failure areas, result in higher source term releases to the environment, and hence higher consequences. Several parameters were evaluated with respect to consequences for the single weather trial in Table 1 and 2 to determine the validity of this supposition.

- source term released to the environment
- peak containment pressure
- maximum leak area
- initial pressure at which containment failure occurs

Inspection of the results finds the following general trends:

- risk is proportional to the source term release to the environment
- more corrosion area and deeper corrosion in most, but not all, cases resulted in higher risk
- lower peak pressures are associated in most, but not all, cases with higher risk
- higher maximum containment failure area in most, but not all, cases resulted in higher risk
- lower initial containment failure pressure in most, but not all, cases resulted in higher risk

A detailed examination of certain cases were made to ascertain the causes for instances when having larger corrosion area (leaks at a lower pressure), not having a higher consequence (Midheight 50% Corrosion 10x vs. Midheight 65% Corrosion 10x for example). Examination of the containment pressure and containment failure area found that initial containment failure occurs earlier for the 65% corrosion case. The early venting of the containment occurs and causes a lower containment pressures later in the accident. This in turn causes the failure area (crack area) later in the accident. The combination of a smaller crack area and lower containment pressure results in less source term being released to the environment and hence a lower consequence. This illustrates that issues other than the extent of corrosion (in this particular case with the initiation time of the failure) impact the consequence results.

In addition, the cases with Midheight 50% and 65% Corrosion show the same risk for both the LTSBO and STSBO cases. This is due to the relatively small difference in the area vs. pressure curves and the MELCOR pressure vs. time curves. However, other simulations performed, which are not presented here, showed similar, though not identical risks between these two conditions.

6 CONCLUSION

This study has examined the effects of local liner degradation in a reinforced concrete containment vessel for a PWR plant on the consequences during a severe accident. These analyses were conducted using the Sandia-developed MELCOR and MACCS codes to perform the accident progression simulations and the resulting consequence calculations, respectively. Five different containment conditions were examined for both the long- and short-term station black-out (LTSBO and STSBO) accident scenarios.

The previous section provides mostly general trends and also shows cases that were observed where the timing of the containment leakage effected to consequences. Higher levels of liner corrosion and the extent of that corrosion lead to leakage at lower pressures and to larger amount of crack area at lower pressures. Though in some cases, small amounts of corrosion were shown to act as a vent whereas to lower the internal pressure of the containment vessel prior to the inventory of radionuclides was fully developed. Therefore, when more consequence significant material was generated within the containment, the pressure could be lower than in a case with no corrosion, or less corrosion.

It is important to closely examine how degradation can change the consequence response of the containment on a case-by-case basis. Due to the more detailed information and fidelity in the different responses, the more deterministic accident scenarios may be more useful when examining a specific case of degradation say in an actual containment vessel. It cannot be stressed enough that the cases of degradation examined here are extremely limited in scope and should not be used to generalize results. There are also a number of assumptions that would require additional consideration when performing a site specific analysis. Due to these assumptions and that the results of this paper are based on hypothetical cases of liner corrosion, the specific results reported in this study should in no way be applied to making specific regulatory decisions on existing plants. The purpose of this work was to demonstrate a potential methodology for use in case specific scenarios.

DISCLAIMER NOTICE

This work was performed under the auspices of the U.S. Nuclear Regulatory Commission (USNRC), Washington, D.C. The findings and opinions expressed in this paper are those of the authors, and do not necessarily reflect the views of the USNRC or Sandia National Laboratories.

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